NON-PUBLIC?: N
ACCESSION #: 9201290022
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Perry Nuclear Power Plant, Unit 1 PAGE: 1 OF 10

DOCKET NUMBER: 05000440

TITLE: Plant Shutdown Due to Circulation Water System Pipe Rupture
EVENT DATE: 12/22/91 LER #: 91-027-00 REPORT DATE: 01/21/92

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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Supervisor, Extension 6185

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: AA COMPONENT: VTV MANUFACTURER: I208
X EA BKR G182
B KE PSX
REPORTABLE NPRDS: Y
N
N

SUPPLEMENTAL REPORT EXPECTED:

ABSTRACT:

On December 22, 1991, at 0205 hours, a manual reactor scram was inserted per Integrated Operating Instruction (IOI) - 8 due to a non-isolable break in the Circulating Water System (N71) piping. The break occurred on a section of 36 inch fiberglass reinforced plastic pipe which supplies cooling water to the auxiliary condensers. At 0259, the shift supervisor declared an Alert based on reports of rising water level received and indications available in the Control Room. All required notifications were made regarding the Alert declaration.

Equipment anomalies and malfunctions which occurred after the manual scram was inserted are summarized in the text of this LER. The cause of

the pipe rupture was attributed to a combination of factors which included a pre-existing flaw in the fiberglass, functional degradation of a pipe support and improper installation of an O-ring gasket at the fiberglass to steel transition flange. The plant was restarted on January 3, 1992 after repairs were made to the affected piping and associated supports. Additionally, the equipment anomalies and malfunctions which occurred after plant shutdown were investigated and corrective actions taken where appropriate.

END OF ABSTRACT

I. Introduction

On December 22, 1991 a fast reactor shutdown was initiated following the catastrophic rupture of a 36 inch condenser circulating water line. At the time of the event, the plant was in Operational Condition 1 (Power Operation) at 100 percent of rated thermal power with the Reactor Pressure Vessel RPV! at approximately 1024 psig and saturated conditions.

II. Description of the Event

(See Appendix A for chronological sequence of events)

At 0138 hours on December 22, 1991, reactor power was increased from 99 to 100 percent power upon completion of a weekly surveillance test. At 0152 hours, an annunciator was received for low circulating water chamber level. At 0154 the Control Room received reports that the motor and diesel fire pumps had started and that the start-up transformer deluge system had initiated. It was also reported at that time, that a large vapor cloud was seen in the vicinity of the Unit 1 start-up transformer. At 0157, Control Room personnel observed that the cooling tower basin was rapidly decreasing and that pump amperage and discharge pressure were oscillating considerably for the existing Circulating Water System (N71) KE! configuration. Decreasing vacuum in the "A" auxiliary condenser was also noted.

At 0200 hours, the Control Room Unit Supervisor (US) ordered a decrease in reactor power to 80 percent. This action was taken with the assumption that the "A" auxiliary condenser could be isolated to stop the suspected system leakage. The Control Room personnel thereafter noticed that vacuum was also decreasing in the "B" auxiliary condenser. There were subsequent reports to the Control

Room of large amounts of water in the transformer yard and Turbine Building. Based upon the above considerations, the US directed entrance into Integrated Operating Instruction
IOI) - 8, "Shutdown
by Manual Reactor Scram." Reactor core flow was reduced and a manual scram was inserted at 0205.

A plant operator later reported to the Control Room that a large leak existed at the 36 inch circulating water pipe inlet to the Heater Bay at the 620 foot elevation. As a result, the US ordered the A and B circulating water pumps secured at 0210 hours. Reactor pressure was being controlled by opening steam bypass valves in accordance with the Plant Emergency Instruction (PEI) - B13, RPV Control. Reactor pressure control was subsequently transferred to the Safety Relief Valves (SRVs). The Reactor Core Isolation Cooling System (RCIC) was also utilized to augment pressure control. Six (6) Level 3 automatic scram signals were initiated due to Level transients experienced while using SRV pressure control. No additional rod motion was experienced as all control rods were fully inserted during the manual scram. At 0224 the "C" circulating water pump was secured.

At 0259, the shift supervisor declared an Alert due to reported rising groundwater level. Although the actual groundwater levels never reached the

height specified by the Emergency Plan for Alert declaration, in the Shift Supervisor's judgment there was sufficient justification for the Emergency Plan activation. The initial NRC notification was made at 0311 hours to report the Alert declaration. Follow-up notifications were made at approximately 1 hour intervals thereafter in addition to telephonic information requested by both Region III and NRR personnel. All required notifications for state and local officials were also made in a timely manner. The Alert was terminated at 1151 on December 22, 1991.

Various equipment malfunctions and anomalies occurred after the manual scram was inserted and are briefly described in Section IV below.

III. Apparent Cause of Pipe Rupture

The catastrophic failure of the 36 inch auxiliary circulating water supply line on December 22, 1991, occurred in a fiberglass elbow in the pipe just prior to the point where the pipe transitions from fiberglass to carbon steel. The pipe was located in a yard area where the pipe exits the ground prior to entering the Heater Bay building.

Several probable causes were evaluated individually and in combination. It is believed that no individual causal factor solely precipitated the pipe failure. An eight (8) inch axial groove on the pipe elbow exterior was determined to be an area of high stress concentration and the primary contributor to the ultimate failure. Additional factors contributing to the failure were the failure of a pipe support to function as an anchor point for the steel piping, which allowed the transfer of undesirable loading stresses to the fiberglass elbow, coupled with incorrect installation of an 0-ring in the transition area between the fiberglass and steel piping, which placed additional stress on the fiberglass elbow. Preventive measures to address identified causal factors were incorporated into the pipe repair process.

IV. Equipment Malfunctions and Anomalies

As previously stated, various equipment problems were experienced after the fast reactor shutdown on December 22, 1991. A brief discussion of the significant occurrences is provided below.

A. Electrical Equipment

1. Bus L11 Failure to Transfer

Upon plant shutdown, i.e., turbine trip, the plant auxiliary loads are transferred to plant startup power sources. This is accomplished automatically by: (1) opening 13.8kV breaker L1102 and closing breaker L1006 and (2) opening 13.8kV breaker L1202 and closing breaker L1009. Both of these breaker automatic transfer schemes are driven by the same relay logic. The L1202 to L1009 transfer properly occurred, and the L1102 and L1006 transfer failed. Upon inspection of 13.8kV breaker

L1006, maintenance found that its closing springs were discharged. All spring charging switches, fuses, etc., were found to be in proper position. Maintenance determined that a subcomponent of the breaker mechanism had broken. The part was replaced and a retest was successfully completed.

Several additional problems occurred as a direct result of the failure of Bus L11 to transfer and were resolved when power was restored to the bus. They are as follows:

a. CRD Pump B tripped due to the momentary de-energization of the "loss of oil pressure" relay.
b. Switch S112 (345kV Main Transformer disconnect switch) would not open due to loss of power to the motor which operates the switch.
c. Various containment isolations occurred due to the loss of Reactor Protection System (RPS) Bus "B" and other low voltage buses:

o Reactor Water Clean Up
o Reactor Water Sample Line
o Backup Hydrogen Purge
o Balance of Plant

d. A Control Room Emergency ventilation recirculation initiation occurred as a result of losing 120 VAC Panel K-1-N.

2. Motor Feed Pump (MFP) Breaker Failure to Close

The MFP breaker logic was set in AUTO-START response mode at the time of the event. With the two Reactor Feed Pumps turbines tripped, the MFP will feed water into the reactor vessel continuously or until a vessel Level 8 is reached. After a short period of time, the operator can reset the Level 8 trip signal and the MFP will again auto start. This trip/reset action occurred 15 times over a two hour period. On the sixteenth trip reset, the MFP did not automatically start.

Subsequent Engineering review of the MFP motor's breaker control logic did not reveal any anomalies which explain the breaker's failure to close on the sixteenth close actuation demand. This review included examination of the breaker's anti-pump control logic.

In addition, the breaker was removed from the cubicle and cycled satisfactorily using the breaker testing equipment. The breaker was disassembled and contacts were inspected. The breaker was reassembled and operated several times in the test position in the switchgear. No problems were found.

3. Startup Transformer Deluge Initiation

This Fire Protection System feature functioned per design when the rate

of rise sensors detected a rapid temperature rise when the comparatively hot N71 water (approximately 80 - 85 degrees Fahrenheit) hit the much cooler transformer. The amount of water and location of water contact did not pose a problem as evidenced by the continuous operation of Startup Transformer 100-PY-B.

4. Equipment Problems Resulting from Water Intrusion

a. Several instruments and a power distribution component in the Emergency Service Water Pumphouse (ESWPH) were damaged by water which entered the building through a series of conduits. This was the only known safety-related equipment affected as a result of flooding. The water which entered the conduit originated in an electrical manhole which became flooded during the pipe rupture event. The affected equipment was repaired or replaced.

b. Several instruments on the non-safety related control rod hydraulic skids became partially submerged from water which entered the Intermediate Building. A walkdown was performed to determine if any other equipment may have been affected based on the maximum height of the water observed in the building. Potentially affected equipment will be meggered and inspected.

B. Mechanical Equipment

1. Scram Discharge Volume Failure to Drain

The scram discharge volume (SDV) failed to drain following the scram due to a failed stem coupling on the outboard drain valve 1C11-F0181. The coupling joins the actuator stem to the disc stem. A notification was made to the NRC at 2225 hours on December 22, 1991 to report the SDV drain valve failure pursuant to the requirements of IE Bulletin No. 80-14. The valve was repaired in accordance with instructions provided in GE Service Information Letter (SIL) 422.

2. Instrument Air Pressure Not Maintained During Event

It was originally believed that a problem existed in the Instrument Air System due to an inability to maintain system pressure above 86 psig with a scram inserted and the Safety Relief Valves being cycled. A detailed evaluation of the sequence of events, system pressure and overall system response was performed. The analysis concluded that the system had functioned as designed during the event and the Unit 1 Instrument Air Compressor was able to supply all required air for important equipment manipulations. The analysis revealed interrelations associated with operating modes of the compressors which were not immediately understood.

C. Structural

The only significant structural damage resulting from this event was confined to the pipe support discussed previously and the soil in the area where the subject piping exited the ground. Some of the soil and stone used around the yard area structures was displaced as a result of the flooding. Additional areas affected included a concrete walkway which was partially damaged and minor housekeeping problems from displaced silt and debris.

V. Safety Analysis

None of the equipment problems or anomalies described impacted equipment required to safely shutdown the plant; therefore, this analysis will focus mainly on the flooding aspects.

The water discharged by the 36" diameter N71 line break located north of the Heater Bay at approximately 620' elevation, generally flooded the yard are in the immediate vicinity of the break. Approximately one to two feet of water could have existed for a short duration at the west boundary of the flooded area.

A. Normal Design Flow Path

Normally, most of the water from the break would be dissipated by surface run-off towards low lying areas away from the plant. (For this break, most of the water would run-off in the north and north-west direction and some in the north-east direction). Some of the water would seep through the class B/C fill (at a very slow rate, as Class B/C fill is nearly impervious) around the building and reach the Underdrain system. The Underdrain system consists of a 1'-0" thick porous concrete mat under the building foundations and a 12" diameter porous pipe routed around the perimeter of the plant. The porous pipe carries the collected water to nine (9) individual pumps located in manholes spaced around the nuclear island. The water collected in the manholes would be pumped to the gravity discharge piping (36" to 48" diameter steel pipe, at E1. 588' high point! to E1. 579' low point!). In the unlikely event of the failure of all nine (9) pumps, the water level in the manholes would rise to E1. 588' and be drained to the ESWPH via the gravity discharge piping. The underdrain system is designed for a postulated break in the circulating water system (12'-0" diameter fiberglass pipe) and is sized to handle the flow from such a break. The break in the 36" diameter pipe which occurred above grade was determined to be bounded by the break postulated for the design basis of the Underdrain system.

B. Estimate of Actual Flow Path

A walk-down conducted on December 22, 1991, revealed that the cover for the manhole #20, immediately to the west of the N71 pipe break, had been left open. This provided a direct and a much more rapid path for some of the flood water to the Underdrain system. This along with the water that seeped

through the ground to the Underdrain system, is considered to be the main flow path to the Underdrain system. The pumping capacity of the Underdrain pumps was exceeded for some time (this explains the high water level alarm received in the Control Room after the break; the alarm is set at E1. 568.5').

The pumped discharge portion of the Underdrain system was probably subjected to a more rapid flow from the break (due to the open manhole) than anticipated by design. However, this did not create a safety concern since the pumped discharge system is not the primary system for keeping the water level below E1. 590'. The Gravity Discharge system, designed to perform this function, has been shown to be adequate to handle a break in the N71 system which envelopes the current break (discussed above). Further, the ground water level was lowered to E1. 568.5' soon after the break as confirmed by a walkdown on December 24, 1991, and piezometer water level readings taken on December 26, 1991. This confirms that the Underdrain system performed its function as designed.

Additionally, due to the open manhole #20, there is a possibility that the capacity of the gravity discharge portion of the Underdrain system was temporarily exceeded. This would result in the water level rising above E1. 590' in the manhole. However, this water would be discharged to the lake via the Gravity Discharge system before it could fill the porous concrete and the Class A fill to E1. 590'. Thus, the water level could not have exceeded E1. 590' (design basis of the Underdrain system).

The path of ingress of water to the plant has been determined to be as follows:

1. Below E1. 590', water most probably entered the safety related buildings through the holes/tears in the waterstops/water proofing membranes at the rattle spaces and piezometer tubes. The amount of in-leakage was also somewhat aggravated for this occurrence by the temporary loss of power to sump pumps within the buildings.

2. Above E1. 590' all the water came into the plant when the electrical manholes filled and water ran back through the duct banks into the plant, into the Service Water pump house and into the ESW pump house. The amount of water intrusion above E1. 590' was insignificant and as such had no safety consequences. The cables in the electrical manholes were specified to operate for forty years submerged in water. The only safety-related equipment was in the ESW pump house where water entered into the building at the south east zone Junction Box JB1-2114. Water then passed through a series of conduits and boxes and ended up in Motor Control Center (MCC) EF1A12 causing the failure of a space heater transformer. Although this had no safety consequences, it is significant because of water which flowed into a safety-related

switchgear. The inlet point for this water has been sealed to prevent any future occurrence.

The extent of in-leakage to plant structures can be attributed to a very rapid entry of flood water into the open manhole, causing the Underdrain system to fill up rapidly. It should also be noted that, for the most part, the floor drains were able to dissipate the water adequately. Thus the items designed to keep the buildings free of water performed in an acceptable manner. The actual flood path for this break was not the path anticipated by design, largely due to the open manhole; however, the systems designed to handle flooding performed adequately as demonstrated by the fact that no essential safety-related equipment was lost as a result of the flooding. Therefore, this event is not considered to be safety significant.

VI. Similar Events

There have been no catastrophic piping failures or similar events for which a plant shutdown was initiated.

VII. Corrective Actions

A. The following corrective actions were developed to address the causal factors associated with N71 pipe rupture described in Section III above.

1. The fiberglass elbow for the Auxiliary Condenser inlet piping was replaced with an identical elbow from Perry Unit 2. The replacement elbow was inspected for potential defects prior to installation and increased in thickness to enhance its pressure capacity. The auxiliary condenser discharge piping elbow will be evaluated to determine the need for any additional reinforcement prior to the end of RF03. An evaluation was performed to justify interim operation.

2. Careful attention was paid to the correct assembly of the material transition flange O-ring to ensure proper flange mate-up. The auxiliary condenser discharge line was inspected and found to have a similar flange mating problem. This line will be reworked to correct the problem prior to the end of RF03. An evaluation was

performed to justify interim operation.

3. A modification was performed to significantly upgrade the supports for the auxiliary condenser inlet and discharge lines. The support modification and the replacement elbow reinforcement were completed prior to starting up the N71 system.

B. The following corrective actions were taken to address the items discussed in Section IV of this LER.

1. As described previously in Section IV.A.1, the breaker subcomponent which caused the failure of Bus L11 to transfer after the scram was replaced and breaker L1006 was successfully retested.

2. As described in Section IV.A.2, the control logic for the Motor Feed Pump (MFP) was reviewed and did not reveal any anomalies. The MFP breaker was disassembled and inspected. The breaker was reassembled and satisfactorily retested. No equipment problem was identified and no further actions were deemed necessary.

3. For the Startup Transformer deluge initiation discussed in Section IV.A.3, no corrective actions were required since the system response to the event was per design.

4. The wetted equipment in the Emergency Service Water (ESW) pumphouse was repaired or replaced as required (see Section IV.A.4.a). Additionally, a modification was performed to seal the conduits which provided the pathway for water intrusion in the ESW pumphouse and add a drainage port for the junction box which filled with water during the event.

5. Affected equipment in the Intermediate Building, described in Section IV.A.4.b, will be inspected and meggered as necessary.

6. With regard to the Scram Discharge Volume (SDV) Drain Valve discussed in Section IV.B.1, a replacement coupling was installed in accordance with the instructions provided in GE Service Information Letter (SIL) 422. Additional instr

ctions were added to associated work order to ensure proper implementation during the installation process.

7. As described in Section IV.B.2, the review of Instrument Air System operation determined that the overall system response during the event was per design. Therefore, no additional actions are required for this item.

8. The soil adjacent to the damaged N71 piping and support was replaced per direction of Engineering department personnel. The remaining structural damage described in Section IV.C was minor in nature and had no effect with regard to plant systems or structures. Cosmetic repairs to the yard area will be required to rake displaced stones and repair a damaged sidewalk. Completion of these items will be prioritized commensurate with ongoing plant activities.

Additionally, all licensed and non-licensed plant operators will receive training on this event as part of requalification training.

Energy Industry Identification System Codes are identified in the test as XX!.

APPENDIX A

CHRONOLOGICAL SEQUENCE OF EVENTS
Circulating Water System Rupture
December 22, 1991

0154 - Automatic start of Diesel Fire Pump and Motor Driven Fire Pump; Indication of Deluge System Initiation on Main Transformer.

0200 - Low Pressure Indications on Circulating Water Pump discharge pressure; Cooling Tower Basin Low Level Alarms; Major rupture identified on Circ Water System and cavitation reported. Operators reduce power to 80%.

0205 - Reduced Recirculation Flow to 52 MLBS/HR and initiated manual Reactor Scram in accordance with IOI-8.

0210 - Secured A and B Circulating Water Pumps.

0224 - Secured C Circulating Water Pump.

0225 - Manually Closed Outboard MSIVs; Established pressure control using Safety Relief Valves. Level maintained using Motor Feed Pump.

0259 - ALERT declared in accordance with Emergency Plan.

0400 - After level 8 trip caused by SRV cycling, MFP failed to restart. RCIC used to maintain RPV level.

0737 - Shutdown Cooling Established using RHR loop A.

1107 - Entered Cold Shutdown.

1151 - Terminated ALERT; Entered Recovery phase.

ATTACHMENT 1 TO 9201290022 PAGE 1 OF 1

CENTERIOR
ENERGY

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January 21, 1992
PY-CEI/NRR-1442 L

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Perry Nuclear Power Plant
Docket No. 50-440
LER 91-027

Dear Sir:

Enclosed is Licensee Event Report 91-027 for the Perry Nuclear Power Plant.

Sincerely,

Michael D. Lyster

MDL:RWG:ss

Enclosure: LER 91-027

cc: NRC Project Manager
NRC Sr. Resident Inspector
NRC Region III

*** END OF DOCUMENT ***